NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
WATERFORD STEAM ELECTRIC STATION UNIT 3	05000 382	1 OF 8

TITLE (4)

INSIDE AND OUTSIDE CONTAINMENT ISOLATION VALVES FAILED LEAKAGE CRITERIA

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
06	11	97	97	021	01	12	16	97	N/A	05000
									N/A	05000

OPERATING MODE (9): 5

POWER LEVEL (10): 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
T.J. GAUDET, LICENSING MANAGER	(504) 739-6666

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
D	KP	ISV	W255	Y					
D	KP	ISV	C283	Y					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On June 11, 1997, at approximately 1100 CDT, while in MODE 5 and at 0% power, leakage rate testing found the inside containment isolation valve for fire protection containment penetration 60 to have failed to meet its leakage acceptance criteria. The outside containment isolation valve for this penetration had been leak tested earlier in the outage and was also found to have failed to meet its acceptance criteria. The unacceptable leakage for both valves has been attributed to excessive corrosion. The outside containment isolation valve was reworked and reinstalled, and the inside valve has been replaced. The as-found condition is being reported per 10CFR50.73(a)(2)(ii). This condition did not pose an actual threat to the health and safety of the public.

REQUIRED NUMBER OF DIGITS/CHARACTERS FOR EACH BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
3	VARIES	PAGE NUMBER
4	UP TO 76	TITLE
5	6 TOTAL 2 PER BLOCK	EVENT DATE
6	7 TOTAL 2 FOR YEAR 3 FOR SEQUENTIAL NUMBER 2 FOR REVISION NUMBER	LER NUMBER
7	6 TOTAL 2 PER BLOCK	REPORT DATE
8	UP TO 18 -- FACILITY NAME 8 TOTAL -- DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

REPORTABLE OCCURRENCE

When the second isolation valve[ISV] for containment penetration[BD] 60 failed to meet its periodic Local Leakage Rate Testing (LLRT) acceptance criteria, Waterford 3 personnel immediately determined the condition to have presented a potential leakage path under postulated accident conditions. Since both valves met their acceptance criteria when tested during the previous refueling outage, it is uncertain when these valves degraded to an unacceptable condition. Reporting guidance usually uses time of discovery for such failures, but this degradation has been conservatively determined to have occurred during the last operating cycle. Because of this and the fact that the failures occurred on the same penetration, a four hour ENS notification was made at approximately 1230 CDT on the event date pursuant to 10CFR50.72(b)(2)(i). This report is being submitted pursuant to 10CFR50.73(a)(2)(ii) as a condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

INITIAL CONDITIONS

At the time of the event, Waterford 3 was in Mode 5, Cold Shutdown, conducting Refueling Outage 8 (RF08) activities. Technical Specification 3/4.6.1 requires that containment integrity be maintained only in Modes 1-4. There were no Technical Specification LCOs in effect, no major equipment out of service and no procedures being performed specific to this event.

EVENT DESCRIPTION

The capability of the containment vessel to maintain design integrity is ensured by a comprehensive design, analysis, and testing program. 10CFR50 Appendix J provides for the periodic leaktight verification of systems and components that penetrate containment, and the establishment of acceptance criteria for such tests are contained in Waterford 3's LLRT program. Containment isolation valves are those which are relied upon to perform a containment isolation function and are typically subjected to Type C LLRT testing on a periodic basis.

Penetration 60 is classified as a class A1 moderate temperature and pressure penetration for the passage of fire protection water[KP] through a 3-inch pipe into the containment vessel. It consists of a pneumatically operated globe valve (FP-601A)[BD-ISV], manufactured by W-K-M Valve Co., model #70-29-1-DRTS, as the outside isolation valve, and a wafer-style check valve (FP-602A)[BD-ISV], manufactured by C&S Valve Co., model # K15EEEY96A, as the inside isolation valve. FP-601A is open during normal and shutdown operational modes, but is automatically closed upon a containment isolation actuation signal or a loss of power.

This fire protection line supplies fire suppression water to fire hose standpipes and sprinkler heads[KP-SRNK] in the containment vessel. It is designed as a dry-line pre-action deluge system. The detection system installed throughout the protected area provides indication to control room operators who manually actuate a deluge valve [KP-INV] located upstream from FP-601A. The system is pressurized to 150 psig from the yard main fire loop up to the deluge valve. The deluge valve admits water to the piping ready to discharge through the standpipes and the sprinklers when their fusible elements open. Supervisory air[LF] pressure at approximately 40 psig is maintained in the piping downstream of the deluge valve to verify integrity of piping and sprinklers. A trouble alarm[LF-PA] sounds if the supervisory pressure is not properly maintained. All piping is carbon steel and, except for that in the containment penetration area, is rated nonsafety and nonseismic.

Type C tests are performed by local pressurization to design basis accident (DBA) containment pressure (Pa) of 44 psig. On April 2, 1997, FP-601A failed its acceptance criteria of 5000 standard cubic centimeters per minute (sccm) when it failed to pressurize to 44 psig. It was noted that about 20 gallons of black water was drained from the normally dry penetration piping, indicating the presence of corrosion products. Condition Report (CR) 97-1104 was written to document the as-found failure, and Work Authorization (WA) 01159463 was prepared to rework the valve and its operator. The valve was repaired and reinstalled, and on June 11, was successfully retested with a measured as-left leakage of 898 sccm. At approximately 1100 CDT the same day, FP-602A failed its acceptance criteria of 10000 sccm when it failed to pressurize to 44 psig in accordance with procedural criteria. CR 97-1456 was written to document the as-found failure of both isolation valves and to initiate the reporting process. At approximately 1230 CDT, a notification was made to the NRC Operations Center to report the as-found failure of Penetration 60. WA 01160817 was prepared to replace the check valve. The valve was replaced and, on June 14, was successfully retested with a measured as-left leakage of 20 sccm, thus restoring the penetration operability.

Prior to its failure in RF08, FP-601A had successfully passed its as-found testing in each refueling outage since RF-2 when it failed due to corrosion/erosion of the valve seat disc. FP-602A had been successfully tested each refueling outage since RF-3 when it failed due to valve seat/disc wear.

CAUSAL FACTORS

The failure of FP-601A and FP-602A was attributed to excessive corrosion and resultant fouling of seating surfaces due to standing water in the penetration piping having not been completely drained. Because this is a dry-line system, the only sources of moisture are from leakage by or actuation of the deluge valve and the moisture present in the station air system, which provides the supervisory air pressurization of the piping. Although no conclusive evidence has been found, the water is believed to have been introduced to the penetration piping by actuation of the deluge valve. The lack of moisture removal capability allowed water to accumulate in the line and corrosion products to form on the valve seating surfaces, which caused the unacceptable leakage.

The line is fitted with a strainer[KP-STR] and drain trap assembly[KP-DRN] between the deluge valve and FP-601A valve to remove accumulated water. Each shift, operators verify no leakage past the deluge valves by pushing the drip check valve on the drain trap. No leakage was reported during the last operating cycle, but during investigations

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related to this event, the strainers were found to be clogged. Station air passes through coalescing type filters[LF-FLT] to remove moisture prior to the fire protection piping. However, the filters have not been periodically changed, nor have the strainers been periodically inspected and cleaned. This oversight has been attributed to inadequate review of a design change. When design change DC-3337 to install the strainer/drain trap assemblies and the coalescing filters in the two fire protection penetrations was completed in November 1992, no positive mechanism was in place to ensure that these preventive maintenance tasks were, in fact, incorporated.

System operating procedure, OP-009-004, "Fire Protection," was determined to be inadequate, such that no specific quidance was given to operators as to the actions necessary to effectively drain the water from the dry piping after an actuation of the deluge valve. The unique configuration of this dry-pipe system was also determined to have contributed to this omission.

CORRECTIVE MEASURES

The penetration piping was drained, the defective valves have been repaired or replaced, and the penetration has been restored to within its leakage criteria. The drain traps and strainers have been cleaned and repaired, and a repetitive maintenance task has been initiated to inspect and clean the strainers and to change the air filters each year.

Procedure NOECP-313, "Design Change Implementation and Closeout," has been revised to include verification of repetitive tasks having been implemented as specified in the design change. A representative sample of design change and modification request packages will be selected and reviewed to determine the extent, if any, of those packages where the repetitive tasks necessary to ensure equipment maintenance and reliability were omitted.

Also, OP-009-004 has been revised to include specific instructions for operators to drain the dry piping for these two penetrations in the event of an actuation of the deluge

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valves. No further water has been found to have accumulated in the penetration piping since the as-left activities.

SAFETY SIGNIFICANCE

The primary reactor containment system and the engineered safety features of the plant ensure that the radiological exposure to the public resulting from a DBA is below the guidelines established in 10CFR100. The containment vessel is designed to withstand the pressure and temperature transients calculated to exist after a DBA(loss of coolant accident (LOCA)). The two fire protection lines penetrate containment in an area of the Reactor Auxiliary Building which is provided with high efficiency filtration and iodine adsorption for the area's ventilation exhaust. This system is referred to as the Controlled Ventilation Area System (CVAS) and maintains a slight negative pressure within the CVAS boundary following a LOCA.

Because the two isolation valves for this penetration were unable to withhold the prescribed Pa, the as-found leakage was unquantifiable. If worst case assumptions are applied to a DBA in this case, the uncontrolled release of radioactive materials from containment through the isolation valves and 3-inch piping to outside atmosphere could exceed 10CFR100 criteria. The potential safety significance is evaluated probabilistically.

For a large release to occur, a LOCA would have to occur during the time when the valves were failed, the LOCA would have to be severe enough to result in containment temperatures high enough to cause the sprinkler heads inside containment to fuse and open, the deluge valve would have to fail open, and the fire protection system piping upstream of the deluge valve and outside the CVAS boundary would have to fail. If this were to occur dose criteria would be exceeded (for example, the exclusion area boundary and control room thyroid doses could be several times higher than the acceptance criteria and the control room skin dose could be about 50% higher). The probability of this release, however, is below 1E-09 per reactor year, which is considered non-risk significant.

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TEXT CONTINUATION

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If the fire protection system piping failed inside the CVAS boundary, almost all of the iodine and particulates would be filtered. In this case, the only significant dose consequences would be the control room skin dose, which might be about 50% higher than acceptable. Since this is a 30 day dose, there would be ample time to take action to isolate the leak and avoid exceeding the skin dose limit. For this scenario to occur, the LOCA would have to occur during the time when the valves were failed and the LOCA would have to be severe enough to result in containment temperatures high enough to cause the sprinkler heads to fuse and open, as in the above scenario, except that the fire protection system piping would have to fail within the CVAS boundary. This scenario is dominated in probability by failure of the piping downstream of the deluge valve, since failure of the piping upstream of the deluge valve would require the additional failure of the deluge valve. The probability of this release is well below 1E-07 per reactor year, which is considered non-risk significant.

Because this condition is non-risk significant, the potential safety significance is considered low.

SIMILAR EVENTS

A review of Waterford 3 Licensee Event Reports submitted since 1995 identified a similar event reported as LER 96-009-01 dated November 21, 1996. This report, in part, documents where like containment isolation valves in redundant instrument penetrations for the Containment Vacuum Relief system[BF] failed to meet leakage testing acceptance criteria due to excessive roughness in the valve bore body.

No similar events during that period were identified where inside and outside containment isolation valves for the same penetration failed to meet their leakage criteria.

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ADDITIONAL INFORMATION

Energy Industry Identification System (EIIS) codes are identified in the text within brackets [ ].